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Experiments and Simulations of ITER-like Plasmas in Alcator C-Mod

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Abstract
Alcator C-Mod is performing ITER-like experiments to benchmark and verify projections to 15 MA ELMy H-mode Inductive ITER discharges. The main focus has been on the transient ramp phases. The plasma current in C-Mod is 1.3 MA and toroidal field is 5.4 T. Both Ohmic and ion cyclotron (ICRF) heated discharges are examined. Plasma current rampup experiments have demonstrated that (ICRF and LH) heating in the rise phase can save volt-seconds (V-s), as was predicted for ITER by simulations, but showed that the ICRF had no effect on the current profile versus Ohmic discharges. Rampdown experiments show an over-current in the Ohmic coil (OH) at the H to L transition, which can be mitigated by remaining in H-mode into the rampdown. Experiments have shown that when the EDA H-mode is preserved well into the rampdown phase, the density and temperature pedestal heights decrease during the plasma current rampdown. Simulations of the full C-Mod discharges have been done with the Tokamak Simulation Code (TSC) and the Coppi-Tang energy transport model is used with modified settings to provide the best fit to the experimental electron temperature profile. Other transport models have been examined also.

1. Introduction

Development and control of ITER discharge scenarios is vital in order to maximize the duration of the flattop phase and minimize the risk of disruption during the rampdown phase. Alcator C-Mod is performing ITER-like experiments to benchmark and verify projections to 15 MA ELMy H-mode Inductive ITER discharges[1,2], and simulations of these discharges are being developed to improve the modeling approaches used for ITER. Several key ITER plasma and discharge parameters are targeted in the C-Mod experiments, including the safety factor q_{95}, density relative to Greenwald n/n_{Gr}, normalized pressure β_N, energy confinement scaling factor $H_{98(y,2)}$, plasma shape and lower single null orientation, large bore startup and early diverting time. The plasma discharge is broken into phases, plasma current rampup, flattop, and rampdown. The main focus has been on the transient ramp phases, with the flattop phase sufficiently long for current profile relaxation. The plasma startup phase has been developed to include a large bore plasma rapidly after breakdown and to divert the plasma as early as possible, about 80 ms out of a 500 ms current rampup time, similar to the 15 s diverting time out of a 100 s current rampup planned for ITER. For the experiments presented, the plasma current in C-Mod is 1.3 MA, toroidal field is 5.4 T, aspect ratio is 3.1, and the plasma elongation is 1.75-1.85.

2. Plasma Current Rampup

The rampup experiments have current rampup times of 500 ms, which are equivalent to a 100 s rampup in ITER, based on the proportionality scaling of $\Delta t_{ramp}/<T_e>^{3/2}a^2$. Both Ohmic and ion cyclotron (ICRF) heated discharges are examined. ICRF heating is initiated at 120 ms early after diverting, and stepped up through the rampup in order to avoid excessive radiation. Shown in Fig. 1 are three rampup cases, Ohmic, 1 MW of ICRF heating and 1-2 MW of ICRF heating. These experiments have demonstrated that ICRF heating in the rise phase can save volt-seconds (V-s) as was predicted for ITER by simulations[2]. The OH1 coil shows a reduced current which implies less V-s are consumed, since this coil dominates the inductive flux swing to the plasma. Shown in Fig. 2 are the V-s components, showing the resistive and inductive internal consumption for the same 3 discharges, from Tokamak Simulation Code (TSC)[2,3] simulations. These V-s savings are resistive and are preserved to the end of the flattop, while V-s associated with current profile modifications, particularly entry to and exit from H-mode, are not preserved. These H-mode current profile effects on the OH1 coil current can be seen as a slowing of the coil current rise just after rampup and a faster rise at the beginning of rampdown. The experiments showed that the ICRF had little effect on the current profile versus Ohmic discharges, while simulations of ITER showed reduction of the internal self-inductance l_i. The current diffusion times, in the rampup phase, are ~ 10-15 s and ~ 45 ms in ITER and C-Mod, respectively, while the neoclassical collisionalities, which contributes to the resistivity, are nearly the same in the two tokamaks. Fig. 2 shows that the TSC simulations also indicate the li values are close for the three discharges. The results are thus consistent with poloidal flux diffusion in combination with a diffusive-convective transport model, as was also used in the ITER simulations.

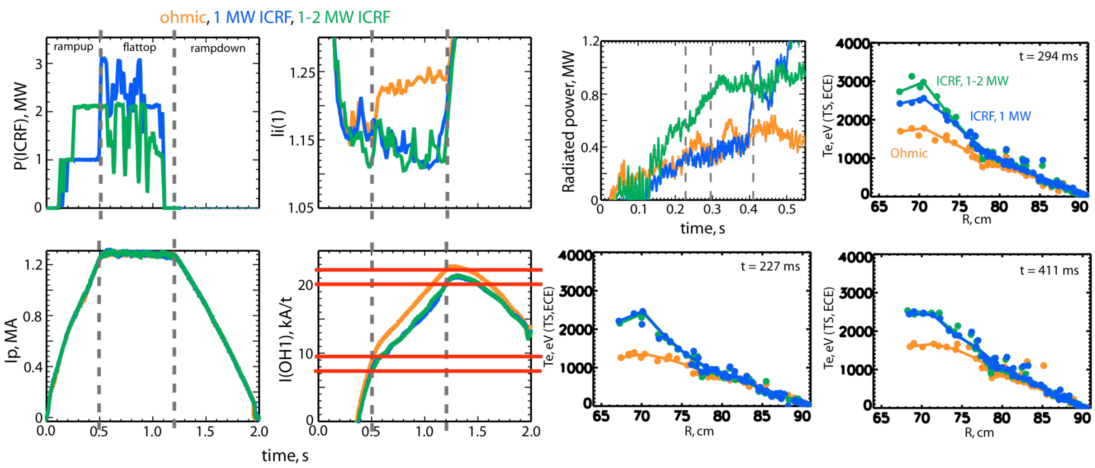

Figure 1. Time histories of the ICRF power, plasma current, li(1), and OH1 coil current for rampup discharges, with only Ohmic heating (orange), 1 MW of ICRF (blue), and 1-2 MW of ICRF (green). V-s are saved according to the OH1 current, but the li values are very similar. Time history of the total plasma radiated powers and electron temperature profiles at 227, 294 and 411 ms, for these rampup discharges showing in spite of varied radiation levels the Te profiles are similar in the outer half of the plasma over the rampup phase.

Experiments indicate that the radiation from the plasma does not appear to contribute to current profile peaking via electron temperature profile peaking. This is shown in Fig. 1 where the radiated power for the ICRF discharges is the same and twice that of the Ohmic discharge, but the l_i values are all very close. Also shown are the electron temperature profiles for the three cases at 227, 294, and 411 ms, which show a very stiff outer profile relative to the core where the heating deposits. The reason for considering this is that the radiated power profile is highest in the outer half of the plasma minor radius, based on bolometric profile measurements[4]. The heating deposition profiles and time histories for the ICRF heating in one of the discharges, calculated in TRANSP with the TORIC full wave module[5], are shown in Fig. 2. The heating scheme is hydrogen minority, which leads to strong hydrogen absorption and subsequent thermal electron heating, which can be as much as twice that on the thermal ions. The heating occurs within a normalized minor radius of 0.30, and increases the central electron temperature but does not appear to broaden the overall profile significantly. The ability to manipulate li with ICRF power and its effectiveness for ITER, will need further examination.

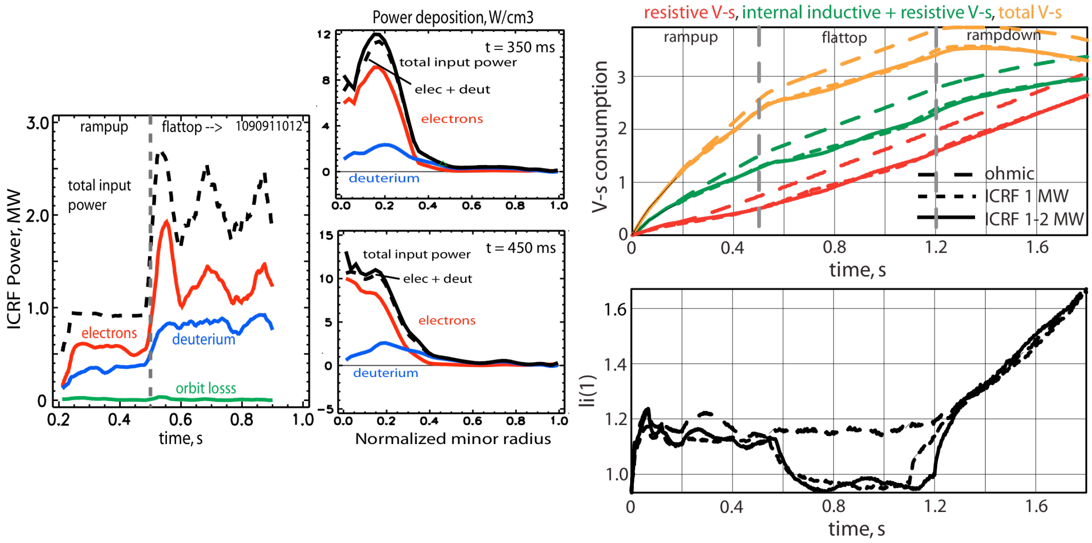

Figure 2. Power time history and deposition profiles, from TRANSP, on H-minority (total input power), thermal electrons and thermal ions at 350 and 450 ms during the rampup discharge with 1 MW of ICRF heating. Time histories of the flux consumption (Axial method) and li(1) from TSC simulations of the three discharges shown in Fig. 1, showing the reduction in resistive V-s for two ICRF heated cases, the effects of entry to and exit from H-mode in the inductive term, and the similar li(1) values in the rampup phase in spite of the ICRF heating.

The effect of the density in the rampup was examined and is shown in Fig. 3, which shows the OH1 coil current, the l_i, line integral density and central electron temperature from ECE measurements, for four Ohmic rampup discharges. Only the lowest density shows a substantial V-s savings, and the l_i values are similar at the end of the rampup. The density and central electron temperature show the expected inverse relationship, and the radiated power at the end of the ramp ranged from 0.25 to 0.55 MW, the higher values at the higher densities. Also shown in Fig. 3 are the electron density and temperature profiles, showing that the temperature is increasing across the entire profile. The sawtooth onset time from the central electron temperature ECE shows a progressive delay with lower density, indicating the current profile is at least temporarily broadened.

Simulations of the rampup phase of ITER with lower hybrid (LH) power injection predict saving significantly more V-s than with ICRF, and a strong modification of the current profile. Initial experiments with LH injection in ITER-like experiments on C-Mod, at a range of densities indicates that 0.4 MW of injected power was effective at saving V-s, and did delay the onset of sawteeth relative to Ohmic and ICRF heated discharges at the lowest and intermediate densities examined.

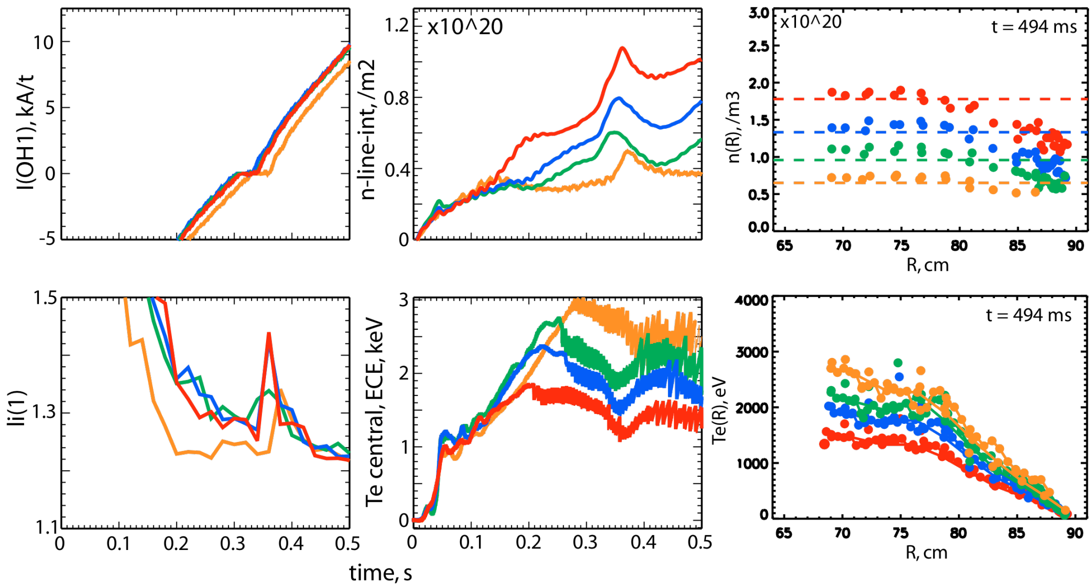

Figure 3. Time histories of the OH1 coil current, li(1), line integral density and central electron temperature during the current rampup phase for 4 densities and Ohmic heating only, showing a noticeable volt-seconds savings only for the lowest density, and a progressive delay in the sawtooth onset as the plasma became hotter, but little effect on li by the end of the rampup. Profiles of the electron density and temperature at the end of the current rampup phase for the same four cases, indicating a uniform temperature increase across the plasma, which should result in a broader current profile.

3. Plasma Current Rampdown

The rampdown phases examined in experiments on C-Mod have durations of 350, 650, and 750 ms, equivalent to ITER rampdown times of 70, 130, and 150 s, respectively. The plasma elongation is reduced during the current rampdown from 1.8 to 1.4 to avoid vertical instabilities, and the plasmas remain diverted to 10% of their flattop plasma current. All cases remain on the midplane coupled to the ICRF antenna. Both ELM-free and EDA H-modes have been sustained in the rampdowns.

Rampdown experiments have shown an over-current in the Ohmic coil (OH1), as was predicted in simulations of ITER rampdown, when the H to L transition occurs at the end of flattop[2]. Since ITER would prefer to operate as close to coil current/field limits as possible, a sudden rise in the central solenoid current at the end of the flattop phase could cause a shutdown. Shown in Fig. 4a are the time histories of the OH1 coil current for 4 rampdown discharges, each with a different H to L transition time. The case that transitions close to the beginning of the rampdown time 1.15 s shows the largest over-current, the next case to transition at 1.45 s shows a weaker over-current, and the next case to transition at 1.65 s shows no apparent over-current. The fourth case, with a faster I_P rampdown rate, has the H to L transition at 1.32 s and a much weaker over-current. The OH coil over-current can be mitigated by remaining in H-mode into the rampdown, or by large power injection in conjunction with L-mode (from simulations). C-Mod experiments show that the H-mode must persist for a sufficient period into the rampdown phase to avoid the OH coil over-current, or the plasma current must be ramped down sufficiently fast to make its effect unimportant.

Experiments have shown that when the EDA H-mode is preserved well into the rampdown phase and ICRF power is present, the density and temperature pedestal heights, and their overall profiles, decrease during the plasma current rampdown, which is desirable to avoid reaching the Greenwald density limit. Having the temperature pedestal decrease in the rampdown is also desirable to avoid significant reverse currents near the plasma edge. Shown in Fig. 4b are the stored energy and line integral density during three discharges, demonstrating a strong reduction that keeps the n/n_{Gr} value approximately constant at the end of flattop value. If the H-mode is in an ELM-free regime during the rampdown, then the density and pedestal parameters tend to suddenly step down in response to reductions in ICRF heating or radiative collapses. This provides important information for modeling the rampdown phase of ITER discharges, and strongly influences the current profile projections. Understanding the power dependence and threshold conditions for ELM regimes in the rampdown will be necessary to guarantee that regulated H-modes persist in ITER's rampdown.

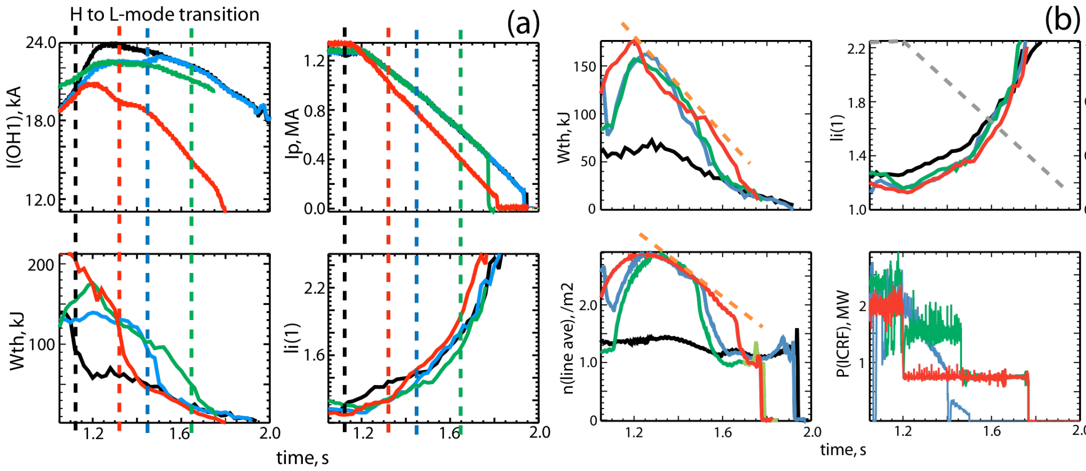

Figure 4. (a) Time histories of the OH1 coil current, plasma stored energy, plasma current and li(1) for three rampdown discharges with the same rampdown rate but different H to L transition times, and one discharge with a faster rampdown rate, showing the OH1 coil over-current at the H to L transition. (b) Time histories of the plasma stored energy, line integral density, li(1), and ICRF power for four rampdown discharges, Ohmic L-mode (black), ICRF H-mode (red, green, and blue) with varying ICRF power, and H to L transition times, showing the strong reduction in density and stored energy associated with the reduction of pedestal temperature and density as Ip is reduced(gray,dashed).

From the three different rampdown rates examined other trends have been identified. The l_i is approximately correlated with the plasma current, that is at the same I_P value the same l_i value is reached. This would not be true if external current drive were used. It is also observed that for a given current rampdown rate there is an L-mode and an H-mode trajectory for $l_i(t)$, when the plasma transitions from H to L-mode it moves from one to the other. The H-mode track can delay the rise in the l_i value by about 0.2 s out of a 0.75 s rampdown, but this is effective only in the first half of the rampdown, after which the trajectories, in L or H-mode, become very similar, since l_i is rising rapidly.

4. Simulations of Experimental Discharges

Predictive simulations of the C-Mod discharges have used the Tokamak Simulation Code (TSC), augmented with the ICRF source deposition from TRANSP interpretive analysis of the discharge. TSC is an axisymmetric free-boundary transport evolution code. Coil currents, plasma current, and toroidal field are taken from the experiment, and feedback systems are used to maintain the overall plasma position and current. Two density profiles are used in these simulations, an L-mode and an H-mode, with the magnitude determined by the experiment, and are shown in Fig. 5. The Coppi-Tang[6] energy transport model is modified, by adjusting the magnitude and profile broadness, to provide the best fit to the experimental electron temperature profile. This model is based on analytic drift wave theory and the ansatz of profile consistency. The original model settings[7] produced a more peaked temperature profile than measured, which made the simulated plasma l_i value too high in the L-mode rampup phase of the discharges, whether Ohmic or ICRF heated. The experimental discharges can be reproduced reasonably well with some modifications to the transport model, and shown in Fig. 5 are some comparisons of the electron temperature profile for an ICRF heated discharge at four time slices in the rampup. The agreement is reasonable over time, although details of the profiles are not fully represented. These simulations have identified the importance of the electron temperature profile in the outer 1/2 to 1/3 of the minor radius in L-mode for properly modeling the current profile evolution. A full discharge simulation is shown in Fig. 6 for an ICRF heated case (1-2 MW), showing some parameter comparisons with experimental measurements. Here the flattop H-mode phase is ELM-free where radiated power is typically higher than EDA H-modes. The evolution at transitions are typically not well modeled, such as the L to H-mode and H to L-mode, as can be seen in the loop voltage plot. The pedestal representation is also difficult since it is in a very small region in C-Mod. Although the pedestal height was matched well to the experiment by suppressing the local thermal diffusivity, the associated li values during the H-mode were lower than the experimental reconstructions.

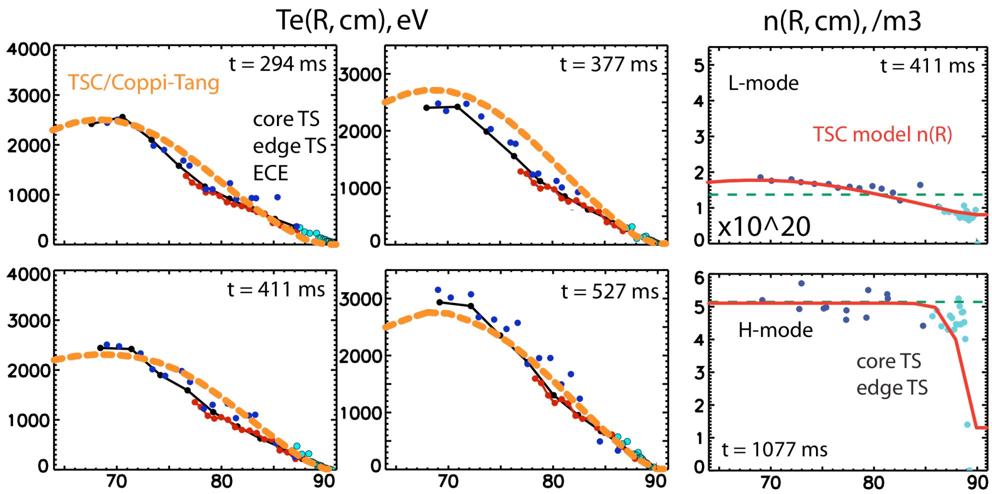

Figure 5. Comparison of measured electron temperature profiles at 4 time slices in the current rampup from Thomson scattering and ECE diagnostics, and the simulated profiles from the Coppi-Tang model in TSC. Although the overall agreement is reasonable the details are not well represented. The model density profile shapes used in the simulations for L and H-mode regimes, in TSC, and their comparisons with experimental profiles at single time slices.

The impurities and associated radiation posed a very difficult problem for modeling, and was circumvented by enforcing the measured total radiation profile and Z_{eff} from the experiment. Shown in Fig. 6 is the power balance and Z_{eff} for an ICRF heated discharge demonstrating the importance of balancing the inputs and losses in order to accurately model the electron channel power balance and understand the power conducted to the plasma edge. The C-Mod ITER-like discharges radiate approximately 50% of their power in flattop and 25-50% in the rampup phase[11], due primarily to the molybdenum impurity, which is close to the radiated power fraction desired in ITER to control the conducted power to the divertor.

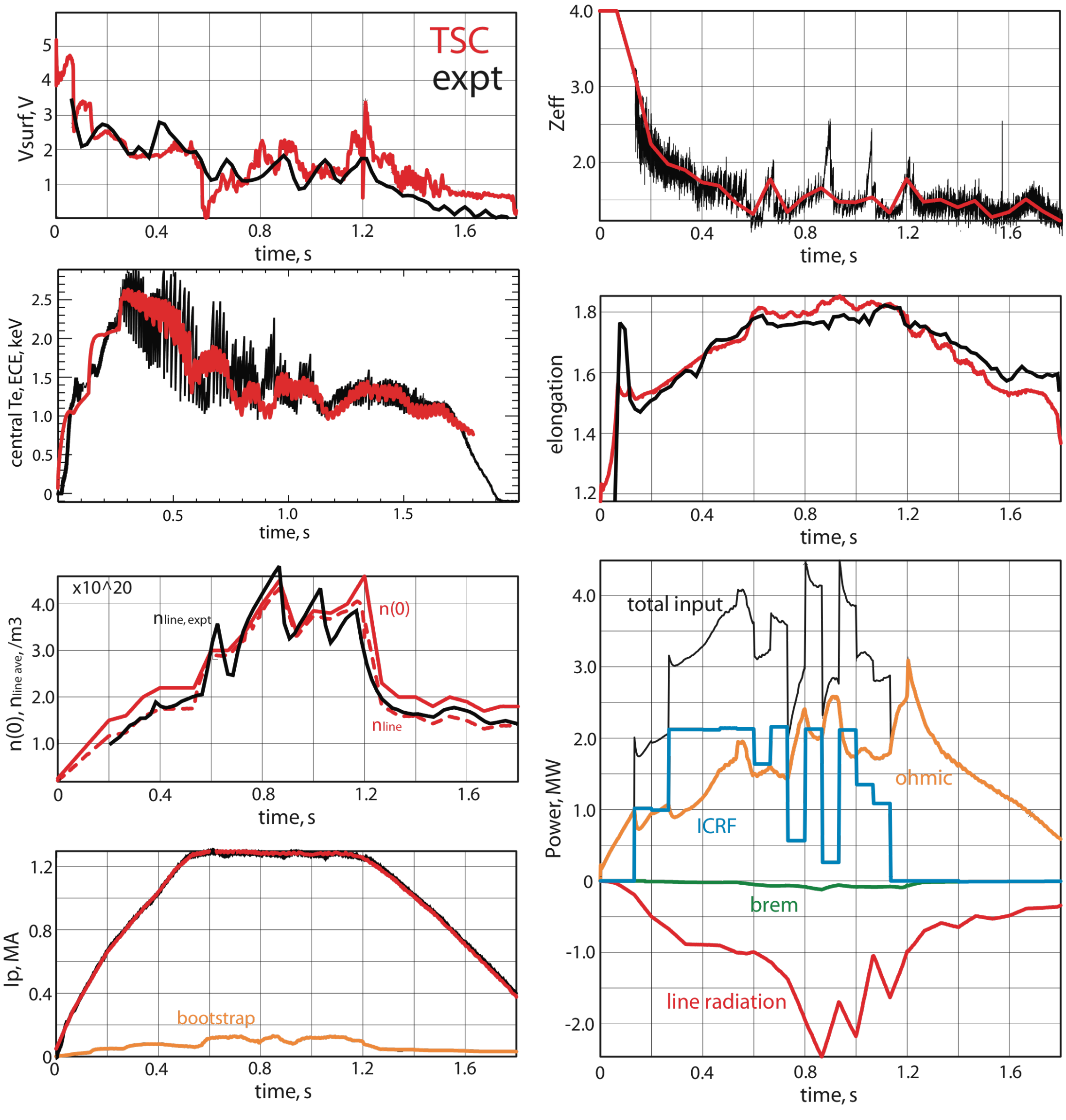

Figure 6. Time histories of surface voltage, central electron temperature, density, plasma current, Zeff, elongation, and input and loss powers for an ICRF heated discharge, comparing a predictive simulation with TSC and experimental data. Differences arise from model deficiencies, time resolution, and inaccurate representation of the evolution at transitions.

The transport model is a critical component in the discharge modeling activity. The GLF23[8] theory based transport model was also tried in the simulations, but did not function properly in the transient low plasma current phase. In order to examine other models the Bohm/gyro-Bohm[9] (BgB) and the CDBM[10] energy transport models are also examined for C-Mod ITER-like discharges in the L-mode rampup phase, with model settings from JET and JT-60U discharges, respectively. The BgB and CDBM models have been derived by comparison of a large number of discharges, while the Coppi-Tang (CT) model has not, and best overall parameters for magnitude and profile broadness in this model are still being identified. Shown in Fig. 7 are the three models applied to an Ohmic discharge rampup, with the experimental values for comparison. All the models over-predict the central temperature very early and then under-predict it later, but ultimately catch up to the data by the end of the ramp. The sawtooth onset times are too early for the BgB and the CDBM indicating overly peaked temperatures, while the CT model is late indicating an overly broad profile. The l_i predictions are reasonable, with the BgB and CDBM models doing well over much of the ramp, and the CT under-predicting due to the broader temperature.

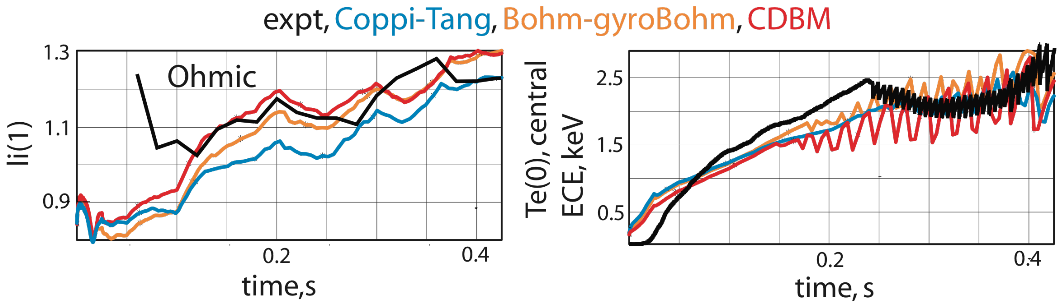

Figure 7. Time histories of li(1) and central electron temperature for Coppi-Tang (blue), Bohm gyro-Bohm (orange), and CDBM (red) transport models versus the experiment.

5. Conclusions

These experiments on Alcator C-Mod and related modeling address the physics and operational basis for ITER inductive discharges. Rampup experiments have examined V-s savings, current profile evolution, early divert and large bore startup, ICRF and LH heating and radiation losses. Rampdown experiments have examined the Ohmic heating coil over-current at the H to L transition, the H-mode pedestal behavior, and current profile evolution. Efforts in discharge modeling will continue with various combinations of experimental data and predictive models to identify the areas that require the most detailed representation and have the strongest influence over the discharge behavior.

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